UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-4005

May 8, 2006

Joseph E. Venable
Vice President Operations
Waterford Steam Electric Station Unit 3
Entergy Operations, Inc.
17265 River Road
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SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3 - NRC PROBLEM IDENTIFICATION AND RESOLUTION INSPECTION REPORT 05000382/2006008

Dear Mr. Venable:

On March 24, 2006, the U. S. Nuclear Regulatory Commission (NRC) completed a team inspection at your Waterford Steam Electric Station, Unit 3. The enclosed report documents the inspection findings, which were discussed with you and other members of your staff during an exit meeting on March 24, 2006.

This inspection was an examination of activities conducted under your license as they relate to the identification and resolution of problems, compliance with the Commission's rules and regulations and the conditions of your operating license. The team reviewed 237 condition reports, apparent cause and root cause analyses, as well as supporting documents. In addition, the team reviewed crosscutting aspects of NRC- and licensee-identified findings and interviewed personnel regarding the safety conscious work environment.

On the basis of the sample selected for review, there were no findings of significance identified during this inspection. The team concluded that, in general, problems were properly identified, evaluated, and corrected. The team concluded that a positive safety-conscious work environment existed at your Waterford Steam Electric Station, Unit 3. Several examples of minor problems were identified, including conditions adverse to quality that were not identified and entered into your corrective action program.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Linda Joy Smith, Chief
Plant Engineering Branch
Division of Reactor Safety

Docket: 50-382
License: NPF-38

Enclosure:
NRC Inspection Report 05000382/2006008
ATTACHMENT A: Supplemental Information
ATTACHMENT B: Waterford 3 Pressurizer Surge Line Temperature Change Rate
ATTACHMENT C: White Paper on Effect of Diesel Sump Pump Inoperability on Ultimate Heat Sink Operability

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Senior Project Engineer, DRP/E **(VGG)**
Team Leader, DRP/TSS **(RLN1)**
RITS Coordinator **(KEG)**
DRS STA **(DAP)**
S. O'Connor, OEDO RIV Coordinator **(SCO)**
ROPreports
WAT Site Secretary **(AHY)**

ADAMS: √ Yes ☐ No Initials: __lјs____
√ Publicly Available ☐ Non-Publicly Available ☐ Sensitive √ Non-Sensitive

DOCUMENT: R_WAT\2006\WT2006-08RP-ELC.wpd

RI:DRP/E	PE:DRP/B	PE:DRP/A	SOE:DRS/OE	RI:DRP/E
ELCrowe/lmb	DHOverland	MABrown	MEMurphy	GFLarkin
/RA/ T	/RA/	/RA/	/RA/	/RA/ T
5/5/06	5/5/06	5/5/06	5/5/06	5/5/06
BC:DRP/E	**SRI:DRS/EB2**	**BC:DRS/EB2**		
DNGraves	D. Proulx	LJSmith		
/RA/	/RA/	/RA/		
5/8/06	5/5/06	5/8/06		

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-382
License:	NPF-38
Report:	05000382/2006008
Licensee:	Entergy Operations, Inc.
Facility:	Waterford Steam Electric Station, Unit 3
Location:	Hwy. 18 Killona, Louisiana
Dates:	March 6-24, 2006
Inspectors:	M. Brown, Project Engineer, Projects Branch A E. Crowe, Resident Inspector, Projects Branch E G. Larkin, Resident Inspector, Projects Branch E M. Murphy, Senior Operations Engineer, Operations Branch D. Overland, Project Engineer, Projects Branch B
Approved by:	L. J. Smith, Chief Engineering Branch 2 Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000382/2006008; Entergy Operations, Inc., 03/06-24/2006; Waterford Steam Electric Station, Unit 3; biennial baseline inspection of the identification and resolution of problems.

The inspection was conducted by two resident inspectors, one senior operations engineer, and two project engineers. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

Identification and Resolution of Problems

- The team reviewed 237 corrective action program documents, apparent and root cause analyses, as well as supporting documents to assess problem identification and resolution activities. Based on this review, the team found the licensee's process to identify, prioritize, evaluate, and correct problems was generally effective; thresholds for identifying issues remained appropriately low and, in most cases, corrective actions were adequate to address conditions adverse to quality. However, a number of issues were identified associated with the proper identification of degraded conditions in the plant. The team reviewed corrective actions associated with these degraded conditions and design issues at Waterford Steam Electric Station, Unit 3, which had crosscutting aspects in the area of problem identification and resolution.

  The team concluded that a positive safety-conscious work environment exists at Waterford Steam Electric Station, Unit 3, based upon interviews conducted with plant personnel. The team determined that employees and contractors feel free to raise safety concerns to their supervision or bring concerns to the employee concerns program.

Inspector-Identified and Self-Revealing Findings

None

REPORT DETAILS

4. OTHER ACTIVITIES (OA)

4OA2 Identification and Resolution of Problems

a. Effectiveness of Problem Identification

(1) Inspection Scope

The inspectors reviewed items selected across four of the seven cornerstones to determine if problems were being properly identified, characterized, and entered into the corrective action program for evaluation and resolution. Specifically, the team's review included a selection of 237 condition reports, equipment walkdowns, review of operator logs, maintenance records, and station quarterly trend reports. The majority of the condition reports were opened and closed since the last NRC problem identification and resolution inspection completed on May 21, 2004. The team also performed a historical review of condition reports written over the last 5 years for the high pressure safety injection system, main feedwater isolation valves, main steam isolation valves, essential chillers, and the emergency diesel generators. The team reviewed a sample of licensee audits and self assessments, trending reports, system health reports, and various other reports and documents related to the problem identification and resolution program. The audit and self-assessment results were compared with the self-revealing and NRC-identified issues to determine the effectiveness of the audits and self assessments.

The team interviewed station personnel and evaluated corrective action documentation to determine the licensee's threshold for identifying problems and entering them into the corrective action program. In addition, in order to assess the licensee's handling of operator experience, the team reviewed the licensee's evaluation of selected industry operating experience reports, including licensee event reports, NRC generic letters, NRC bulletins, and NRC information notices, and generic vendor notifications to assess if issues applicable to Waterford Steam Electric Station, Unit 3, were appropriately addressed.

A listing of specific documents reviewed during the inspection is included in the attachment to this report.

(2) Assessment

The team determined that, in general, problems were adequately identified and entered into the corrective action program, as evidenced by the relatively few findings identified during the assessment period. The licensee's threshold for entering issues into the corrective action program was appropriately low. However, the team found two examples of ineffective problem identification during this inspection. The licensee also failed in some instances to identify or document deficiencies, which resulted in NRC noncited violations.

Current Issues

Example 1: The licensee failed to identify multiple temperature changes of the pressurizer surge line, which exceeded the heatup and cooldown rate described in Section 5.4.3.1 of the station's Final Safety Analysis Report. Specifically, the inspection team discovered during a plant shutdown in August 2005 that the pressurizer surge line had experienced 19 changes in temperature, which exceeded this limit. This example is further described in Section 4OA2.e of this report.

Example 2: The team found the licensee's identification of adverse trends to be weak. The inspection team reviewed 17 conditions reports, in which the licensee documented inadequacies in the procurement of replacement parts for the station. The licensee had identified a trend of improper parts passing through the receipt inspection, but failed to identify adverse trends related to lack of engineering involvement, as required by the procurement process; failure to perform professional engineering evaluations for parts transferred into the system; and receipt inspection documents missing required attributes. These procurement process weaknesses resulted in a nonseismically qualified synchronization switch being installed in an otherwise operable emergency diesel generator and a nonconforming fuel oil nipple passing receipt inspection.

Example 3: The NRC identified that the licensee missed several opportunities to identify the containment fan cooler condensate flow switches that did not meet the design requirements for detecting a one gallon per minute reactor coolant system leak (NRC Inspection Report 05000382/2005005-01).

Example 4: Control room operators missed several opportunities over a 32.5 hour period to identify that a vacuum had been drawn on the reactor coolant system during refueling outage draindown conditions (self-revealing, NRC Inspection Report 05000382/2005010-03).

Historical Issue

Example: The NRC identified the licensee failed to identify an inappropriate value of the unfiltered in-leakage parameter used to calculate the control room operator dose for design basis accident conditions involving radiological releases (NRC Inspection Report 05000382/2004006-01).

b. Prioritization and Evaluation of Issues

(1) Inspection Scope

The team reviewed condition reports, engineering operability evaluations, and operations operability determinations to assess the licensee's ability to evaluate the importance of the conditions adverse to quality. The team reviewed a sample of condition reports, failure mode analyses, apparent cause and root cause analyses, to ascertain whether the licensee identified and considered the full extent of conditions, generic implications, common causes, and previous occurrences. The team also observed management oversight of the significant conditions adverse to quality, including one Corrective Action Review Board meeting.

In addition, the inspectors reviewed licensee evaluations of selected industry operating experience reports, including licensee event reports, NRC generic letters, NRC bulletins, NRC information notices, and generic vendor notices to assess whether issues applicable to Waterford Steam Electric Station, Unit 3, were appropriately addressed. The team performed a historical review of condition reports covering the last 5 years regarding the high pressure safety injection system, the emergency diesel generators, main feedwater isolation valves, essential chillers, and the dry cooling tower to determine if the licensee had appropriately addressed long-standing issues and those that might be age dependent.

A listing of specific documents reviewed during the inspection is included in the attachment to this report.

(2) Assessment

The team concluded that problems were generally prioritized and evaluated in accordance with the licensee's corrective action program guidance and NRC requirements. The team found that for the sample of root cause analyses reviewed, that the licensee was generally self critical and exhaustive in its research into the history of significant conditions adverse to quality. However, the team found one example of ineffective problem evaluation during this inspection.

Current Issues

Example 1: The inspectors discovered the licensee had categorized the failure of a fuel oil pipe nipple in the Emergency Diesel Generator B in 2002, as a condition adverse to quality. The licensee followed Procedure EN-LI-102, "Corrective Action Process," Revision 4, in making the determination of significance. The inspectors followed the steps of Procedure EN-LI-102 and arrived at the same level of significance, however, the procedure provides a provision for the Condition Review Group to change the level of significance, as warranted by the conditions. The inspectors determined that this was a significant condition adverse to quality because the failure rendered one emergency diesel inoperable. The Emergency Diesel Generator A experienced a failure of its corresponding fuel oil nipple in 2005. The licensee determined this failure was a significant condition adverse to quality solely because of the repetitive nature of the failure.

c. Effectiveness of Corrective Actions

(1) Inspection Scope

The team reviewed 237 condition reports to verify that corrective actions related to the issues were identified and implemented in a timely manner commensurate with safety, including corrective actions to address common cause or generic concerns. The team reviewed corrective actions planned and implemented by the licensee and sampled specific technical issues to determine whether adequate decisions related to structure, system, and component operability were made.

In addition, the team reviewed a sample of those condition reports written to address NRC inspection findings to ensure that the corrective actions adequately addressed the issues as described in the inspection report writeups. The team also reviewed a sample of corrective actions closed to other condition reports and programs, such as work and engineering work requests to ensure that the condition described was adequately addressed and corrected.

A listing of specific documents reviewed during the inspection is included in the attachment to this report.

(2) Assessment

The effectiveness of identified corrective actions to address adverse conditions was generally adequate. The team evaluated several occurrences where the licensee did not effectively address conditions adverse to quality and corrective actions taken were untimely and inappropriate. These included five examples, one identified by the team and four by other NRC inspections, where the licensee failed to take prompt corrective actions to resolve long-standing issues. The team also evaluated nine other findings identified by the NRC baseline inspection program and other NRC inspections at Waterford Steam Electric Station, Unit 3, since the last problem identification and resolution inspection that had crosscutting aspects related to prompt and effective corrective actions to resolve conditions adverse to quality.

Current Issues

Example 1: The reactor coolant draindown procedure failed to identify that temporary vent rigs, required by procedure to properly establish vent paths, included in-line ball valves in series with the vent path and also failed to direct those ball valves be opened to establish the vent path. The licensee was aware of and did not fix the procedure to address the ball valves in 2002 (NRC Inspection Report 05000382/2005010-02).

Example 2: The NRC identified the licensee failed to correct the condition which resulted in multiple cycle timer failures in the essential chiller (NRC Inspection Report 05000382/2005002-01).

Example 3: The NRC identified the licensee failed to prevent recurrence of through wall pipe leakage on the main steam line Pipe 2MS2-123. This deficiency resulted in an unisolable steam leak requiring NRC approval to deviate from the American Society of Mechanical Engineers Boiler and Pressure Code Case N523-2 to perform temporary repairs preventing a plant shutdown (NRC Inspection Report 05000382/2005004-03).

Historical Issues

Example 1: The NRC identified the licensee failed to correct a known deficient condition involving the failure to account for instrument uncertainty to satisfy Technical Specification Surveillance Requirement 4.7.6.5.a. This failure potentially affects the ability of the control room envelope to perform its design function with respect to protecting operators from postulated design basis accidents resulting in radiological releases (NRC Inspection Report 05000382/2004006-03).

Example 2: The NRC identified the licensee failed to correct a known deficient condition involving multiple occasions of accumulator overpressure conditions resulting from degraded hydraulic fluid adversely affecting the main feedwater isolation valve hydraulic actuator pressure relief system. These over pressure conditions potentially result in valve closure stroke times outside design basis values (NRC Inspection Report 05000382/2004005-03).

Example 3: The NRC identified the licensee failed to promptly correct instances where the main feedwater isolation valve actuator thermal relief valves failed to properly function. In one case, the licensee failed to properly address system operability and, for a 2-week period, actual valve operability was unknown (NRC Inspection Report 05000382/2004006-02).

Example 4: The NRC identified the licensee failed to correct deficiencies in the emergency diesel generator loading and fuel oil consumption analysis. The licensee inappropriately closed a corrective action requiring the revisions, which subsequently resulted in the failure to maintain design control of the emergency diesel generator fuel oil storage inventory requirements to ensure a 7-day postaccident fuel oil inventory (NRC Inspection Report 05000382/2004002-05).

Example 5: The NRC identified the licensee failed to determine the cause and precluded recurrence of main steam isolation solenoid-operated dump valve failures. The inspectors noted that the licensee's apparent cause did not provide an extent of condition analysis for the solenoid-operated valve failure (NRC Inspection Report 05000382/2004004-03).

Example 6: The NRC identified the licensee failed to take adequate corrective action to ensure the torque applied to the flow control valve for Accumulator B of main feedwater isolation Valve 1 was sufficient to prevent an o-ring from extruding, resulting in a loss-of-system hydraulic fluid and rendering the valve inoperable (NRC Inspection Report 05000382/2004008-02).

Example 7: The NRC identified the licensee failed on multiple occasions to correct a known deficient condition involving the failure to account for instrument uncertainty to satisfy Technical Specification Surveillance Requirement 4.7.6.5.a. This failure potentially affects the ability of the control room envelope to perform its design function with respect to protecting operators from postulated design basis accidents resulting in radiological releases (NRC Inspection Report 05000382/2004006-03).

Example 8: The licensee failed to replace known age-degraded o-rings affecting the main feedwater isolation valves in the Year 2000 resulting in o-ring failure and inoperability of the Train A feedwater isolation valve on December 27, 2003 (NRC Inspection Report 05000382/2004002-01).

Example 9: The NRC identified the licensee failed to establish appropriate torque specification to ensure adequate o-ring compression that ultimately led to an o-ring failure and the inoperability of the Train A main feedwater isolation valve. The licensee had previously identified concerns related to inadequate work instructions for performing maintenance activities on the main feedwater isolation valves (NRC Inspection Report 05000382/2004002-02).

d. Assessment of Safety-Conscious Work Environment

(1) Inspection Scope

The team interviewed 24 individuals from the licensee's staff, representing a cross section of functional organizations and supervisory and nonsupervisory personnel. These interviews assessed whether conditions existed that would challenge the establishment of a safety-conscious work environment. The team interviewed the site employee concerns program coordinator.

(2) Assessment

The team concluded that a positive safety-conscious work environment exists at Waterford Steam Electric Station, Unit 3. Based on interviews, station personnel felt free to enter issues into the corrective action program, raise safety concerns with their supervision, to the employee concerns program, and to the NRC. The team determined that the majority of safety concerns were addressed through the site's normal chain of command by the relatively few safety concerns entered into the employee concerns program and the small number of allegations made to the NRC.

e. Specific Issues Identified During this Inspection

(1) Inspection Scope

During this assessment, the team performed the inspections scoped in Sections 4OA2 a.(1), 4OA2 b.(1), 4OA2 c.(1), and 4OA2 d.(1) above.

(2) Finding Details

(i) Unresolved Item: 05000382/2006008-01, "Failure to Maintain Design Control of the Pressurizer Surge Line"

Introduction. The team identified an unresolved item related to compliance with 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to translate design-basis heatup and cooldown rates for the pressurizer surge line into appropriate specifications, procedures, and instructions. As a result, Entergy Operations, Inc., failed to effectively control and evaluate pressurizer surge line temperature changes on numerous occasions.

Description. Final Safety Analysis Report (FSAR) Section 5.4.3.1, “Reactor Coolant Piping Design Basis,” and Section 5.4.10.1, “Pressurizer Design Basis,” states, in part, that during heatup and cooldown of the plant, the allowable rate of temperature change for the surge line is limited to 200°F/hr. Technical Requirements Manual (TRM), Section 3.4.8.2, “Pressurizer Heatup/Cooldown,” specifies the limiting condition for operation, in part, as a maximum heatup rate of 200°F per hour and a maximum cooldown rate of 135°F per hour.

On April 18, 2005, Entergy Condition Report CR-WF3-2005-1392 stated that a pressurizer surge line temperature transient occurred with the surge line temperature dropping from 425°F to 140°F, a change of approximately 285°F with approximately 200°F occurring within 8 minutes. Technical Requirements Manual, Section 3.4.8.2 Action specifies, "With any of the pressurizer limits in excess of the above, the operators must restore the affected parameter to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; and enter TRM LCO 3.0.3."

The team noted that Entergy Operations, Inc., failed to restore pressurizer/surge line limits within 30 minutes and perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer/surge line. The team reviewed Entergy Operations, Inc.’s operating procedures for plant heatup and cooldown activities, OP-010-005, “Plant Shutdown,” and OP-010-003, “Plant Startup,” and did not find procedure steps to limit surge line temperature changes to less than 200°F/hr, nor were there any procedure steps to assess whether surge line stress or fatigue limits had been exceeded. This appeared to be a violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” for the failure to translate design-basis heatup and cooldown rates for the pressurizer surge line into appropriate specifications, procedures, and instructions.

The design limit in Report CEN-387-P was based, in part, by temperature gradients greater than 200°F occurring less than 3.6 occurrences per heatup/cooldown cycle for 500 heatup/cooldown cycles over the 40-year life of the plant. Calculation CN-OA-04-53 documented 19 instances where pressurizer insurges, in excess of the volume of the surge line, occurred with a temperature gradient greater than 200°F. These pressurizer insurges occurred during five refueling outage heatup/cooldown cycles (Refueling Outages 8-12)for an average of 3.8 temperature gradients greater than 200°F per heatup/cooldown cycle.

Entergy Operations, Inc. disagreed and provided a paper (Attachment B), which documented their position. While they acknowledged that the FSAR was not up to date, they stated that the pressurizer surge line temperature transient on April 18, 2005, was bounded by Combustion Engineering Owners Group Report CEN-387-P, “Pressurizer Surge Line Flow Stratification Evaluation,” submitted to the NRC in response to NRC Bulletin 88-11, “Pressurizer Surge Line Thermal Stratification.” Report CEN-387-P concluded that the pressurizer surge line met all applicable design codes, FSAR, and other regulatory commitments for the licensed life of the plant considering the phenomenon of thermal stratification in fatigue and stress evaluations. The team noted that this conclusion was based on operating the plant consistent with the assumptions in the evaluation (Report CEN-387-P). Additional inspection is required to complete the review of Entergy Operations, Inc.'s, position and determine whether the licensee was operating their facility within the assumptions of the analysis.

Analysis. The significance of this issue depends on whether or not the analysis bounds past plant operation.

Enforcement. The potential failure to translate the design basis into appropriate specifications, procedures, and instructions to effectively control and evaluate surge line temperature changes, during plant heatup and cooldown, that exceeded those limits described in the FSAR and the TRM is unresolved: (URI 05000382/2006008-01); "Failure to Maintain Design Control of the Pressurizer Surge Line."

(ii) Unresolved Item 05000382/2006008-02, “Failure to Ensure that Written Procedures Adequately Incorporate Regulatory Requirements and Design Basis”

Introduction. The team identified an unresolved item related to compliance with Technical Specification, Section 6.8.1, for the failure to ensure that written procedures adequately incorporate regulatory requirements and the design basis for the dry cooling tower diesel-driven sump pumps.

Description. Waterford Safety Evaluation Report, Supplement 4, Section 2.4.2.3, discusses the design basis rainfall event and combination of events. This supplement commits the licensee to the probable maximum precipitation event. Because of the fact that the motor-driven sumps are not seismically qualified, the NRC requested the licensee analyze the effects of a standard project storm, which consists of 50 percent of the probable maximum precipation event concurrent with an operating basis earthquake. The results of the licensee’s analysis showed the licensee was susceptible to ponding in the dry cooling tower sumps, assuming the loss of all motor-driven pumps, which would endanger the safety-related transformers and motor control centers located in the cooling tower areas.

The licensee submitted Amendment 34, dated January 1984, subsequent to Safety Evaluation Report, Supplement 4. Section 2.4.2.3.4 of this amendment submittal contains an analysis showing the probability of standard project storm and operating basis earthquake is 3.6E-08, which is considered negligible. However, the licensee proposed to provide a 100 gpm portable pump that would be sufficient to pump down the dry cooling tower sumps in the event of the standard project storm. The NRC determined that the portable pump was sufficient (as evidenced in Safety Evaluation Report, Supplement 4) provided the pump was placed in operation within 6 hours. In 2000, after determining that more sump pumping capacity was needed, the licensee installed a diesel-driven sump pump, with 300 gpm capacity, in each dry cooling tower sump. The Design Basis Calculation EC-M99-010 analyzed for a probable maximum precipation event, concurrent with a loss-of-offsite power, and determined that a higher capacity portable pump was needed. The calculation also analyzed for a rainfall equivalent to 60 percent of the probable maximum precipitation event, concurrent with a loss of all motor-driven sump pumps, and determined that a 300 gpm portable pump would be sufficient. The licensee's Procedure OP-100-014, "Technical Specifications and Requirements Compliance," Revision 14, states that two motor-driven sump pumps or one motor-driven pump and one diesel-driven pump are required for ultimate heat sink operability. This procedure implies that the diesel driven sump pump can be out of service indefinitely without affecting operability of the ultimate heat sink. The NRC staff believes this procedure does not adequately address the requirement of the portable sump pump in the design basis of the ultimate heat sink, nor does the procedure require any compensatory actions be taken in the event the diesel-driven sump pump becomes inoperable. Also, the staff believes the controls and location of the diesel-driven sump pump are not adequately addressed by the licensee.

Analysis. The significance of this issue has not been determined.

Enforcement. The licensee has provided a position paper (Attachment C) related to the design basis requirements for the dry cooling tower diesel-driven sump pumps, which has not been fully reviewed by the NRC. The potential failure to ensure regulatory requirements for these pumps is unresolved: (URI 05000382/2006008-02) "Failure to Translate Design Control into Station Documents Regarding Diesel-driven Dry Cooling Tower Sump Pumps"

4OA6 Exit Meeting

The team discussed the findings of the Problem Identification and Resolution inspection with Mr. J. Venable, Vice President Operations, and other members of the licensee's staff on March 24, 2006. Licensee management did not identify any materials examined during the inspection as proprietary.

The licensee acknowledged the findings presented. The inspectors noted that while proprietary information was reviewed, none would be included in this report.

ATTACHMENT A: Supplemental Information
ATTACHMENT B: Waterford 3 Pressurizer Surge Line Temperature Change Rate
ATTACHMENT C: White Paper on Effect of Diesel Sump Pump Inoperability on Ultimate Heat Sink Operability

KEY POINTS OF CONTACT

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R. Fletcher, Training Manager
C. Fugate, Assistant Operations Manager
J. Hall, Operations Training Supervisor - Operator Requalification
J. Holman, Manager, Nuclear Engineering
J. Laque, Manager, Maintenance
R. Murillo, Senior Staff Engineer
R. Osborne, Manager, Engineering Programs and Components
A. Pilutti, Manager, Radiation Protection
O. Pipkins, Senior Licensing Engineer
R. Porter, Superintendent, Mechanical Maintenance
B. Proctor, Systems Engineering Manager
J. Rachal, Design Engineering Supervisor
J. Ridgel, Manager, Corrective Action Program
T. Tankersley, Acting Director, Nuclear Safety Assurance
K. Walsh, General Manager, Plant Operations
B. Williams, Engineering Director
J. Venable, Site Vice President, Waterford 3

NRC

M. Hay, Senior Resident Inspector Waterford 3

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000382/2006008-01	URI	Failure to Maintain Design Control of the Pressurizer Surge Line (Section 4OA2 e.)
05000382/2006008-02	URI	Failure to Translate Design Control into Station Documents Regarding Diesel-driven Dry Cooling Tower Sump Pumps (Section 4OA2 e.)

LIST OF DOCUMENTS REVIEWED

Plant Procedures

NAME	TITLE	REVISION
CEP-IST-1	IST Bases Document	3
EN-OP-115	Conduct of Operations	0
LI-102	Corrective Action Process	4
LI-19645	Quality Related Administrative Procedure	2
MM-006-119	Yard Oil Separator to CW Temporary Pumping System	0
OI-042-000	Watch Station Procedures	1
OP-001-003	Reactor Coolant System Draindown	23
OP-005-004	Main Steam	12
OP-009-008	Safety Injection System	18
OP-100-001	Operations Standards and Management Expectations	22
OP-100-009	Control of Valves and Breakers	17
OP-100-0014	Technical Specification and Technical Requirements Compliance	13
UNT-005-004	Temporary Alteration Control	16

Engineering Reports

ER-W3-2002-0055 ER-W3-2004-0537 ER-W3-2005-0426 ER-W3-00-0337
ER-W3-2003-0010 ER-W3-2005-0305 ER-W3-2002-0278

Calculations

CN-OA-04-53 EC-M99-010 MN(Q)-6-27

Root Cause Analysis Reports for CR-WF3-

2001-0317 2002-0339 2003-0062 2003-3891 2004-759 2004-1011

Condition Reports, CR-WF3-

1997-1227	2004-0759	2004-2404	2005-0109	2005-1626	2005-3831
2000-0441	2004-0821	2004-2487	2005-0132	2005-1646	2005-3840
2000-1347	2004-0835	2004-2496	2005-0134	2005-1694	2005-3872
2000-1455	2004-0865	2004-2517	2005-0197	2005-1821	2005-3872
2001-0596	2004-0903	2004-2520	2005-0217	2005-1836	2005-3902
2001-0673	2004-1011	2004-2522	2005-0346	2005-2070	2005-3914
2001-0782	2004-1047	2004-2545	2005-0413	2005-2139	2005-3924
2001-1284	2004-1190	2004-2547	2005-0415	2005-2267	2005-3928
2001-1367	2004-1208	2004-2549	2005-0471	2005-2272	2005-3960
2002-0468	2004-1312	2004-2638	2005-0489	2005-2350	2005-3961
2002-0470	2004-1340	2004-2690	2005-0530	2005-2402	2005-3985
2002-0588	2004-1446	2004-2722	2005-0587	2005-2469	2005-4038
2002-0678	2004-1480	2004-2734	2005-0590	2005-2489	2005-4065
2002-1410	2004-1518	2004-2766	2005-0591	2005-2536	2005-4066
2002-1842	2004-1553	2004-2884	2005-0592	2005-2546	2005-4067
2002-2799	2004-1572	2004-2890	2005-0608	2005-2548	2005-4147
2003-0147	2004-1593	2004-2928	2005-0717	2005-2600	2005-4149
2003-0577	2004-1621	2004-2973	2005-0763	2005-2679	2005-4151
2003-1192	2004-1645	2004-2995	2005-0804	2005-2685	2005-4173
2003-1202	2004-1646	2004-3066	2005-0805	2005-2695	2005-4251
2003-2758	2004-1668	2004-3130	2005-0806	2005-2780	2005-4444
2003-2759	2004-1679	2004-3200	2005-0839	2005-2799	2005-4480
2003-2991	2004-1684	2004-3219	2005-0852	2005-2819	2005-4597
2003-3088	2004-1716	2004-3244	2005-0921	2005-2837	2005-4647
2003-3649	2004-1751	2004-3413	2005-0966	2005-2844	2005-4694
2003-3891	2004-1753	2004-3460	2005-0967	2005-2869	2005-4915
2004-0251	2004-1763	2004-3464	2005-1132	2005-2874	2005-4917
2004-0304	2004-1810	2004-3695	2005-1143	2005-2990	2005-4929
2004-0309	2004-1850	2004-3720	2005-1173	2005-3006	2005-5024
2004-0326	2004-1854	2004-3725	2005-1247	2005-3091	2006-0006
2004-0420	2004-1855	2004-3753	2005-1260	2005-3293	2006-0058
2004-0464	2004-1863	2004-3853	2005-1279	2005-3308	2006-0164
2004-0483	2004-1880	2004-3881	2005-1315	2005-3455	2006-0200
2004-0494	2004-1942	2004-3924	2005-1332	2005-3474	2006-0380
2004-0508	2004-2002	2004-3944	2005-1346	2005-3659	2006-0492
2004-0634	2004-2228	2004-3949	2005-1362	2005-3698	2006-0759
2004-0651	2004-2290	2004-4000	2005-1363	2005-3812	2006-0767
2004-0701	2004-2320	2005-0033	2005-1392	2005-3822	2006-0839
2004-0703	2004-2326	2005-0081	2005-1463	2005-3830	2006-0895
2004-0721	2004-2382	2005-0098			

Learning Organization Conditions Reports

LO-OPX-2004-0247	LO-OPX-2005-0100	LO-OPX-2005-0217	LO-OPX-2006-0011
LO-OPX-2005-0036	LO-OPX-2005-0103	LO-OPX-2005-0243	LO-OPX-2006-0034
LO-OPX-2005-0085	LO-OPX-2005-0132	LO-OPX-2005-0252	

Work Orders

51697	52825	72604	412565	4599901
51699	57759	72606	4281801	5100331101
52824	62641			

Maintenance Action Items

420105 438981

Miscellaneous Documents

Commercial Grade Evaluation 01214
C-PAC-002
L-19645
L-23993
MMR Project 53465
PO WPY20583
2004 Second Quarter Waterford Quarterly Trend Report
2004 Third Quarter Waterford Quarterly Trend Report
2004 Fourth Quarter Waterford Quarterly Trend Report
Quality Assurance Audit Report QA-12-20050-WF3-1
Quality Assurance Audit Report QA-12-20050-WF3-009
Quality Assurance Audit Report QA-12-20050-WF3-1
PO 10083675

INITIAL MATERIAL REQUEST

INITIAL INFORMATION REQUEST FROM WATERFORD 3
FOR PI&R INSPECTION (Report Number 05000382/2004006)

The inspection will cover the period of October 2002 to March 2004. The information may be provided in either electronic or paper media or a combination thereof. Information provided in electronic media may be in the form of CDs, or 3-½ inch floppy disks. The agency's text editing software is Corel WordPerfect 8, Presentations, and Quattro Pro; however, we have document viewing capability for MS Word, Excel, Power Point, and Adobe Acrobat (.pdf) text files.

Please provide the following information to Peter Alter by March 29, 2004 at the Resident Inspector Office at Waterford-3

All procedures governing or applying to the corrective action program, including the processing of information regarding generic communications and industry operating experiences

Procedures and descriptions of any informal systems, used by engineering, operations, maintenance, security, training, and emergency planning for issues below the threshold of the formal corrective action program

A searchable table of all corrective action documents (condition reports) that were initiated or closed during the period, include condition report number, description of issue and significance classification

Either annotate on the above list or a separate list of all condition reports associated with:

(1) Human performance issues
(2) Emergency preparedness issues
(3) Response to 10 CFR Part 21 reports

A separate list of all condition reports closed to other programs, such as maintenance action items/work orders, engineering requests, etc.

A copy of each significant event review team report and root cause analysis report for the period (not necessarily the whole condition report)

Copies of condition reports (for the period) associated with nonescalated (no response required) or noncited violations for the period

Copies of condition reports for the period associated with repetitive problems or issues

Copies of condition reports for the period associated with ineffective or untimely corrective actions

List of all self assessments or quality assurance assessments/audits for the period

All corrective action program reports or metrics used for tracking effectiveness of the corrective action program for the period

All quality assurance audits and surveillances, and functional self assessments of corrective action activities completed for the period

Control room logs for the Year 2003

Security event logs for the year 2003

Radiation protection event logs for the year 2003

List of risk significant systems from W3 PRA/PSA, based on risk achievement worth (RAW) and "0% availability CDF"

Searchable list of all maintenance action items/work orders for the period

List of all SSC's placed in or removed from the maintenance rule a(1) category for the period

All corrective action documents related to the following industry operating experience generic communications:

NRC Bulletins

NRC Bulletin 2002-001, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity"

NRC Information Notices

NRC Information Notice 2004-001, "Auxiliary Feedwater Pump Recirculation Line Orifice Fouling - Potential Common Cause Failure"

NRC Information Notice 2003-019, "Unanalyzed Condition of Reactor Coolant Pump Seal Leakoff Line During Postulated Fire Scenarios or Station Blackout"

NRC Information Notice 2003-013, "Steam Generator Tube Degradation at Diablo Canyon"

NRC Information Notice 2003-011, "Leakage Found on Bottom-Mounted Instrumentation Nozzles"

NRC Information Notice 2003-008, "Potential Flooding Through Unsealed Concrete Floor Cracks"

NRC Information Notice 2003-005, "Failure to Detect Freespan Cracks in PWR Steam Generator Tubes"

NRC Information Notice 2003-002, "Recent Experience With Reactor Coolant System Leakage And Boric Acid Corrosion"

NRC Information Notice 2002-034, "Failure of Safety-Related Circuit Breaker External Auxiliary Switches at Columbia Generating Station"

Information Request 1 - January 2006
Waterford PIR Inspection (IP 71152; Inspection Report 50-382/06-08)

The inspection will cover the period of March 1, 2004 to March 1, 2006. All requested information should be limited to this period unless otherwise specified. The information may be provided in either electronic or paper media of a combination of this media. Information provided in electronic media may be in the form of e-mail attachment(s), CDs, or 3 ½ inch floppy disks. The agency's text editing software is Corel WordPerfect 10, Presentations, and Quattro Pro; however, we have document viewing capability for MS Word, Excel, PowerPoint, and Adobe Acrobat (pdf.) text files.

Please provide the following by February 8, 2006, to:

U.S. Nuclear Regulatory Commission
Resident Inspector's Office - Attn. Grant Larkin
Waterford Steam Electric Station Unit 3
Entergy Operations, Inc.
17265 River Road
Killona, Louisiana 70066

Note: On summary lists please include a description of problem, status, initiating date, and owner organization

1. Summary list of all condition reports opened during the period

2. Summary list of all open condition reports with significance of "B" or greater which were generated during the period

3. Summary list of all condition reports with significance of "B" or greater closed during the specified period

4. Summary list of all condition reports which were down-graded or up-graded in significance during the period

5. A list of all corrective action documents that subsume or "roll-up" one or more smaller issues for the period

6. List of all root cause analyses completed during the period

7. List of all apparent cause analyses completed during the period

8. List of root cause analyses planned, but not complete at end of the period

9. List of plant safety issues raised or addressed by the employee concerns program during the period

10. List of action items generated or addressed by the plant safety review committees during the period

11. Summary list of operator work-arounds, engineering review requests and/or operability evaluations, temporary modifications, safety system deficiencies, and control room deficiencies

12. All quality assurance audits and surveillances of corrective action activities completed during the period

13. A list of all quality assurance audits and surveillances scheduled for completion during the period, but which were not completed

14. All corrective action activity reports, functional area self-assessments, and non-NRC third party assessments completed during the period

15. Corrective action performance trending/tracking information generated during the period and broken down by functional organization

16. Current procedures/policies/guidelines for:
    1. Condition Reporting
    2. Corrective Action Program
    3. Root Cause Evaluation/Determination
    4. Deficiency Reporting and Resolution

17. A listing of all external events evaluated for applicability at Waterford during the period

18. Condition Reports or other actions generated for each of the items below [ADAMS accession numbers or other cross reference listed for some]:

    1. Part 21 Reports (2005-41-00; 2005-38-00 [ml053180299]; 2005-37-00; 2005-33-01 [ml052860229]; 2005-30-01 [ml052640220]; 2005-26-01 [ml052910389]; 2005-22-00; 2005-20-00; 2005-17-00 [ml051110087]; 2005-16-00 [ml051100285]; 2005-13-00 [ml050950428]; 2005-12-01 [ml052080368]; 2005-12-00 [ml050630275]; 2005-10-00 [ml050560142]; 2005-07-00; 2005-05-01 [ml051100355]; 2005-01-00 [ml043520077]; 2004-27-01 [ml043280541]; 2004-24-00 [ml042470299]; 2004-22-00 [ml042660175]; 2004-21-00 [ml042520048]; 2004-17-00 [ml041900058]; 2004-15-00; 2004-14-00; 2004-10-00 [ml041140335]; 2004-08-00 [ml041110893]; 2004-02-01 [ml040420567]

    2. NRC Information Notices 05-32; 05-31; 05-30; 05-29; 05-26; 05-25; 05-24; 05-23; 05-21; 05-19; 05-16; 05-11; 05-09; 05-08; 05-06; 05-02; 04-021; 04-019; 04-016; 04-012; 04-011; 04-010; 04-009; 04-008; 04-007; 04-001

    3. All LERs issued by Waterford during the period

    4. NCVs and Violations issued to Waterford during the period

19. Safeguards event logs for the period

20. Radiation protection event logs

21. Current system health reports or similar information

22. Current predictive performance summary reports or similar information

23. Corrective action effectiveness review reports generated during the period

ATTACHMENT B

Waterford 3 Pressurizer Surge Line Temperature Change Rate

Waterford 3 Pressurizer Surge Line Temperature Change Rate

Purpose

This Paper is to document the Entergy position on the potential NCV of 10CFR50 Appendix B, Criterion III, “Design Control” for not translating design basis criteria into plant operating procedures. The design basis criteria in question is a statement in the FSAR (Section 5.4.3.1) which states:

> *During heatup and cooldown of the plant, the allowable rate of temperature change for the surge line is increased to 200°F/hr as a design requirement specified in Subsection 3.9.1.1.*

Background

The following is a time line of the Entergy response to NRC Bulletin No. 88-11. This concludes that the fatigue life of the Waterford 3 surge line is 40 years which the NRC concurred with.

- The NRC issued NRC Bulletin No. 88-11, Pressurizer Surge Line Thermal Stratification, on December 20, 1988. The purpose of the Bulletin was to request that addressees establish and implement a program to confirm pressurizer surge line integrity in view of the occurrence of thermal stratification and to inform the staff of the actions taken to resolve this issue.
- CEN-387-P was transmitted to the NRC on July 27, 1989. This documented that the Waterford 3 surge line fatigue life is longer than 40 years.
- On August 28, 1989, Entergy sent a letter to the NRC stating that Bulletin 88-11 item 1b, 1c and 1d were addressed in CEN-387-P and that item 1a (visual inspection of the pressurizer surge line) would be addressed during the next refueling outage.
- On March 7, 1990, Entergy sent a letter to the NRC which addressed the results of the visual inspections of the pressurizer surge line. The letter concluded that the Waterford 3 surge line was structurally sound.
- On August 15, 1990, the NRC issued a letter stating there was not enough information in the CEN document to conclude that the pressurizer surge line meets all appropriate Code limits for a 40 year plant life.
- On December 20, 1991, CEN-387-P, Revision 1-P was sent to the NRC to address the concerns of the August 15, 1990 NRC evaluation of the CEN document.
- On May 5, 1992, Entergy sent a letter to the NRC documenting the submittal of the revised CEN document and stated the only remaining action to complete the response to the Bulletin is for the Waterford 3 to update the pressurizer surge line design documentation. This was committed to be completed within 180 days of issuance of a favorable SER by the NRC.

- On June 22, 1993, the NRC issued an SER for CEN-387-P, Revision 1. It was concluded that the analysis in the CEN adequately demonstrates that the bounding surge line and nozzles meet ASME Code stress and fatigue requirements for the 40 year design life of the facility considering the phenomenon of thermal stratification and thermal stripping. The staff requested Entergy to provide a final status of the Waterford 3 activities required by NRC Bulletin 88-11.
- On December 23, 1993, Entergy sent a letter to the NRC stating that all design documents had been updated and that all actions required by NRC Bulletin 88-11 had been completed.

CEN-387-P, Revision 1, is the Combustion Engineering response to NRC Bulletin 88-11. This document addresses pressurizer surge line flow stratification. The document provides a detailed fatigue analysis of stress due to stratified temperature profiles of the fluid in the pressurizer surge line. Note that this document indicates that thermal stratification is assumed for all surge flow as the velocities will always be low. This document also specifically indicates that the stratified temperature analysis envelopes high velocity flow and thermal shock.

The following paragraphs are excerpted from the Thermal Striping Analysis for the pressurizer surge line in CEN-387-P, Revision 1. The conclusion is the "effect of thermal striping is negligible and will not affect the fatigue life of the pressurizer surge line".

> *The term "striping" refers to the thermal oscillations that occur at the hot-cold interface.*
>
> *The period of oscillations was chosen to be 1 second and 4 seconds for the surge line analysis. Test data was measured or was empirically determined to be in the range of ≤ 1 second to 10 seconds. For the large temperature differences and high heat transfer coefficient used in this analysis, the period is closer to 1 second than 4 seconds. A longer period would yield a lower heat transfer coefficient, and therefore smaller changes in metal temperatures. However, to be conservative, the same heat transfer coefficient was used for all cases.*
>
> *The stresses due to each gradient as a function of time were calculated using formulas in ASME Code Section III, NB-3653.2. Table 3.5.3-2 lists the alternating stress calculated for each of the four transients used for evaluating fatigue. As can be seen from this table only one of the four transients contributes anything to fatigue. That transient is number four (4) with an alternating stress of 15,780 psi and a number of allowable cycles of 1.42E7.*

Waterford 3 Design Specification 9270-PE-140 is the project specification for reactor coolant pipe and fittings. This document provides a summary of the design analysis for surge line temperature transients. It includes text sections and 2 tables as they apply to the surge line and surge line nozzle. The tables address temperature differences anticipated as a result of thermal stratification. Table 4.5.15.3.1 lists expected occurrences of temperature differences between the pressurizer and the RCS hot leg and provides the number of expected occurrences. Table 4.5.15.3.2 lists expected occurrences of temperature differences between the top and bottom of the surge line piping. These temperatures differences are for the pressurizer surge line piping and not the fluid temperature in the piping. The number of occurrences is the expected number for the life of the plant.

Entergy Position

The Entergy position is that pressurizer surge line temperature is not required to be specifically monitored per procedure to ensure the design limits are maintained, and that FSAR Section 5.4.3.1 should have been revised in 1993 when the Waterford 3 stress and fatigue analyses and design specifications were revised per NRC Bulletin 88-11 to reflect the results of CEN-387-P, Revision 1. This section of the FSAR has not been revised since the initial FSAR. CR-WF3-2006-0839 was initiated to revise the FSAR. The reasoning for Entergy's position is documented in the paragraphs below.

The pressurizer surge line temperatures during heatup and cooldown are maintained by ensuring the heatup and cooldown limits in the RCS and pressurizer are maintained. The RCS limits are located in the TS and the pressurizer limits are located in the TRM. Temperature changes in the surge line can be greater than 200°F due to thermal stratification and thermal stripping. CEN-387-P, Revision 1 documented that the pressurizer surge line meets Code stress and fatigue requirements for the 40 year design life of the facility considering the phenomenon of thermal stratification and thermal stripping. Analysis in the CEN has indicated that temperature differences of up to 340°F have been evaluated for.

The data recorded by the temperature element in the surge line has shown periods of temperature changes greater than 200°F/hr. Thermal stratification is applicable to all of these recorded temperature changes. These temperature changes do not necessarily reflect the average temperature change of the surge line but reflects a change in local fluid temperature at the temperature element. This recorded temperature changes over time are not the same delta temperatures listed in the tables in 9270-PE-140.

Therefore, the temperature difference in the pressurizer surge line is bounded by the analysis performed in CEN-387-P, Revision 1 and monitoring pressurizer surge line temperature per procedure during heatup and cooldown is not necessary.

Additional Information

The additional information specifically addresses the difference between the surge line temperature increase seen during Refuel 13 and during the shutdown for Hurricane Katrina, and the delta temperature values in 9720-PE-140. It also addresses the reason Waterford 3 does not currently monitor surge line temperature during heatups and cooldowns.

There following information is clarification regarding cycles listed in Design Specification 9270-PE-140 and the temperatures recorded in PI with the temperature element located in the surge line. The graphical data recording the single surge line temperature element over time for our Refuel 13 outage and the Hurricane Katrina outage indicates periods of temperature changes greater than 200 degrees within one hour. Thermal stratification is applicable to all of these recorded temperature changes. Thermal stratification temperature changes were addressed by CEN-327-P (NRC accepted response to NRC Bulletin 88-11). This single temperature element does not necessarily reflect the average temperature change of the surge line but reflects a change in local fluid temperature at the temperature element. The recorded temperature changes of a single point over time is not the same delta temperatures listed in the tables of the W3 Design specification of RCS Piping and Fitting document (document #9270-PE-140). The table 4.5.15.3.1 lists expected occurrences of temperature differences between two different locations; the pressurizer and the RCS hot leg and provides the number of expected occurrences. Table 4.5.15.3.2 lists expected occurrences of temperature differences between the top and bottom of the surge line piping. These tables clearly state this information at the end of their respective sections. Thus comparing a graph of temperature changes with respect to time to these tables is not appropriate.

The effects on the Pressurizer Surge Line due to thermal stratification and thermal stripping were evaluated in CEN-327-P, Revision 1. This was reviewed by the NRC and in the SER the Staff concluded that the surge line meets ASME Code stress and fatigue requirements for the 40-year design life. Waterford 3 currently monitors heatups and cooldowns of the RCS and Pressurizer. The effects of these heatups and cooldowns on the pressurizer surge line have been evaluated in CEN-327-P.

ATTACHMENT C

White Paper on Effect of Diesel Sump Pump Inoperability on Ultimate Heat Sink Operability

1.0 Purpose

This paper provides an answer to the question, what is the original licensing basis for flood protection of essential equipment in the Dry Cooling Tower Areas? The paper also provides the chronology of regulatory requirements and licensing bases that support the conclusion.

2.0 Conclusion Regarding Licensing Basis

The original licensing basis for essential equipment in the Dry Cooling Tower areas is that essential equipment be protected from Standard Project Storm (SPS).

The elements of the licensing basis are the following:

§ The SPS, with all installed sump pumps inoperative, was analyzed as an event less severe than the probable maximum precipitation.

§ Provisions are required to be in place for emplacing the portable sump pump within 6 hours of an SPS event to ensure that the ponding level from SPS does not adversely affect essential equipment if installed pumps are inoperative.

§ The electric pumps are seismically designed but not seismically qualified; therefore they were assumed not to be available following an OBE.

§ The probability of the occurrence of an SPS and OBE is 3.6E-8 and negligible.

In essence, the original licensing basis required that the portable sump pump be emplaced and started within 6 hours of the start of an SPS (sump high level alarm) to ensure that essential equipment in the DCT areas is not flooded.

On July 26, 1999, Condition Report CR-WF3-1999-0789 was initiated to identify that the Dry Cooling Tower sump pump capacities were not sufficient to meet the original licensing basis.

A new discharge path for the DCT sump pumps was installed via DCP-3251. The DCP also replaced the 1 portable sump pump that had a capacity of 100 gpm with 2 portable sump pumps having a capacity of 300 gpm each. The installed sump pump's capacities were reduced from 325 gpm to 270 gpm due to the new piping configuration. The revised time frame for starting the portable sump pump to ensure essential equipment is not flooded was re-established as 3 hours from the start of an SPS (sump high level alarm). Procedure OP-901-521 instructs Operations to operate the DCT Portable Sump Pumps in accordance with OP-003-024, Sump Pump Operation within 3 hours of the sump level alarm.

3.0 Chronology

Regulatory Guide 1.70, Revision 2, September 1975

Waterford 3 is committed to Regulatory Guide 1.70, Revision 2, as noted in section 1.8 of FSAR. Neither Regulatory Guide Section 2.4.2.3, “Effects of Local Intense Precipitation,” or Section 2.4.3.1, “Probable Maximum Precipitation (PMP),” have any requirement to consider OBE or SPS concurrently.

Regulatory Guide Section 2.4.2.3 states:

“Describe the effects of local probable maximum precipitation (see Section 2.4.3.1) on adjacent drainage areas and site drainage systems, including drainage from the roofs of structures. Summarize the design criteria for site drainage facilities and provide analyses that demonstrate the capability of site drainage facilities to prevent flooding of safety related facilities resulting from local probable maximum precipitation.”

The fundamental requirement in the Regulatory Guide is that the applicant ensures that safety related equipment is not adversely impacted from maximum precipitation.

Regulatory Guide 1.59, Revision 2, August 1977

Waterford 3 is committed to Regulatory Guide 1.59, Revision 2, as noted in section 1.8 of FSAR. Regulatory Guide 1.59, Revision 2, does not have a specific requirement to consider OBE and SPS concurrently.

Two important requirements are discussed in the Regulatory Guide.

First, seismically induced floods are associated with land features specific to each site such as streams, estuaries, dam failures, and landslides. This requirement does not apply to flooding in the DCT sump areas.

Second, the Regulatory Guide states that the most severe flood conditions may not indicate potential threats to safety related systems that might result from combination of flood conditions thought to be less severe. The Regulatory Guide states that reasonable combinations of less severe flood conditions should be considered to the extent needed. The Regulatory Guide states that such combinations should be evaluated in cases where the **probability** of their existing at the same time and having significant consequences is at lease comparable to that associated with the most severe hydro-meteorological or seismically induced flood. We judge that the requirement to consider the SPS originates from this requirement. Also, since the probability of a SPS and OBE concurrent was later established to be negligible, we judge that not considering the SPS concurrent with the OBE is in conformance with the Regulatory Guide.

Standard Review Plan 2.4.3, Revision 2 July 1981

Standard Review Plan 2.4.3 does not have a specific requirement to consider OBE and SPS concurrently.

Standard Review Plan 2.4.3, Section I, states:

Included is a review of the details of site drainage…, including the roofs of safety related structures, resulting from potential PMP probable maximum precipitation…"

Standard Review Plan 2.4.3, Section IV, states:

"The local PMF resulting from the estimated local PMP was found not to cause flooding of safety related facilities, since the site drainage system will be capable of functioning adequately during such a storm."

The fundamental requirement in the Standard Review Plan is that the applicant ensures that safety related equipment is not adversely impacted form maximum precipitation.

NRC Safety Evaluation Report, July 1981

The NRC evaluates the effects of a 6-hr duration PMP on the open cooling tower areas and adjacent roofs. The NRC concludes that, assuming one sump pump in each area is inoperable and that the roof drainage system is clogged with debris during the PMP, that the ponding could inundate the transformers and MCC's in the cooling tower areas.

The Safety Evaluation Report makes no reference to SPS or OBE.

FSAR Amendment 25, January 1982

FSAR Section 2.4.2.3.4 was initially added to the FSAR; previously it did not exist. This FSAR Section is titled, "Effects of Standard Project Storm (SPS) on Cooling Tower Areas".

Two important aspects of the licensing basis are established in this FSAR Section.

First, a probability evaluation is documented establishing that the occurrence of an SPS and OBE is 3.6 E-8 and negligible.

Second, FSAR Section states that the SPS was still analyzed, assuming inoperability of all pumps, in order to determine the time available before levels are reached that could affect essential equipment in the Cooling Tower Areas.

Safety Evaluation Report, Supplement 4, October 1982

The SER states the following:

"An alternative combination which should be considered is an operating basis earthquake (OBE), which fails the sump pumps, coincident with a rainfall event less than the PMP. This combination is considered appropriate since the pumps are not seismically qualified[1], and thus cannot be shown to be operable following a seismic event. The staff therefore, requested that the applicant provide an analysis of the effects of a standard project storm (SPS)[2] assuming all four pumps in the cooling tower areas are inoperable."

The SER further states:

"…the staff considered a SPS of 96 hours duration. This event would produce a total rainfall of about 23 inches and would result in a ponding depth of about 1.9 ft in the cooling tower areas assuming that all four pumps are inoperable. Since this is higher than the maximum allowable ponding depth of 1.71 feet, the applicant has proposed to provide a portable pump with a pumping capacity of 100 gpm and sufficient head to pump over the cooling tower wall. …a provision will be included for emplacing the portable pump within 6 hours of a seismic event if the installed pumps fail."

FSAR Amendment 33, September 1983

FSAR Amendment 33 revises Section 2.4.2.3.4 to state the following:

"The maximum height to which rainwater can rise in this area before essential equipment is reached is 1.71 ft (see subsection 2.4.2.3.3d). As shown in Table 2.4-6c, this level would not be reached for over seven hours into the SPS."

"Furthermore, a portable pump is provided, with a pumping capacity of 100 gpm and sufficient head to pump over the cooling tower wall. Provisions are included for emplacing the portable pump within six hours of a seismic event if the installed pumps fail and heavy rains are expected."

Thus, the FSAR Amendment 33 is in agreement with NRC SER Supplement 4 in that the fundamental requirement is to protect essential equipment in the cooling tower areas in the event of a SPS. The specific requirement in FSAR Amendment 33 is that provisions be made for emplacing the portable sump pump within 6 hours of a SPS event and that essential equipment be protected, by ensuring that the ponding level does not reach 1.71 ft. The seismic event is a vehicle to postulate the installed pumps are not available; however, important to the licensing basis is the condition that the electric sump pumps will not be available and that essential equipment needs to be protected prior to the ponding level reaching 1.71 ft.

NRC Letter dated December 18 1984, Issuance of Five Percent Power License,

The NRC issues five percent power license, and Section 2.B.2 of the license approves operation as described in FSAR as supplemented and amended through Amendment 36.

NRC Letter dated March 16, 1985, Issuance of 100% Power License

The NRC issues 100 percent power license, and Section 2.B.2 of the license approves operation as described in FSAR as supplemented and amended through Amendment 36.

Design Change, July 26, 1999

On July 26, 1999, Condition Report CR-WF3-1999-0789 was initiated to identify that the Dry Cooling Tower sump pump capacities were not sufficient to meet the original licensing basis.

A new discharge path for the DCT sump pumps was installed via DCP-3251. The DCP also replaced the 1 portable sump pump that had a capacity of 100 gpm with 2 portable pumps having a capacity of 300 gpm each. The installed sump pump's capacities were reduced from 325 gpm to 270 gpm due to the new piping configuration. The revised time frame for ensuring essential equipment is not flooded was re-established as 3 hours from the start of SPS (sump high level alarm). Procedure OP-901-521 instructs Operations to operate the DCT Portable Sump Pumps in accordance with OP-003-024, Sump Pump Operation within 3 hours of the sump level alarm.